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FPL Energy.

Duane Arnold Energy Center

June 1, 2007

NG-07-0468
10 CFR 50.73

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Licensee Event Report #2007-007-00

Please find attached the subject Licensee Event Report (LER) submitted in accordance with 10 CFR 50.73. This letter makes no new commitments or changes to any existing commitments.

Rob Anderson for

Gary Van Middlesworth
Site Vice President, Duane Arnold Energy Center
FPL Energy Duane Arnold, LLC

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Resident Inspector, DAEC, USNRC

IE22
NRC/NRR

NRC FORM 366 (6-2004) **U.S. NUCLEAR REGULATORY COMMISSION**

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Reactor Scram Due to 1A2 Non-essential Bus Lockout

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	02	2007	2007	- 7 -	0	06	01	2007	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 98

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Bob Murrell, Engineering Analyst	(319) 851-7900

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED

○ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ◉ NO

15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On April 2, 2007, while operating at 98% power, during performance of planned preventative maintenance on Non-essential Electrical 4160V AC Bus 1A2, a bus lockout occured at 11:25. The loss of the 1A2 switchgear resulted in the loss of 'B' Reactor Feed Pump and 'B' Condensate Pump. At 11:25, a manual reactor scram was inserted due to Reactor Pressure Vessel level approaching 170 inches (automatic scram level). The scram went to completion. After the scram, vessel level rose to 211 inches, causing the 'A' Reactor Feed Pump to trip. Before the 'A' Reactor Feed Pump could be restarted and adequate feed flow established, another scram signal was received when reactor level dropped below 170 inches. Control rods were already inserted and isolations had been completed prior to this second scram signal being received. The Reactor Operator successfully restored level using the Feedwater Regulating Valve as directed by the Operations Shift Manager.

The cause of this event is the failure to recognize the risk associated with performing the maintenance on-line without putting physical and process protections in place.

There were no actual safety consequences and no effect on public health and safety as a result of this event.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. Description of Event:

On April 2, 2007, Work Orders (WO) 1135564, "Clean and Inspect three 1A2 Bus Auxiliary 151-201 Over Current (OC) Relays" and 1135565 "Clean and inspect 1A2 Bus 151N-201 Ground Relay" were signed on by the Work Control Center.

One Electrician and one Electrician Apprentice began work shortly after 0800 by removing the relays from panel in 1A2. The relays were taken to the corridor outside the Essential Switchgear room and calibrated. The as-found relay calibration results were found to be within specification.

Both Electricians and a Nuclear Oversight (NOS) observer entered the 1A2 (4160 volt non-essential AC Bus) Non-essential Switchgear room and began replacing the relays, starting with the 151N-201 Ground Relay.

The relay panel case and internal panel contacts were inspected for foreign material prior to re-installation. The Ground Relay was replaced followed by the replacement of its relay contact paddle and finally its cover. The Phase 1 over current relay (151-201) replacement was completed in an identical manner. All actions were completed as required with no abnormalities noticed.

The same actions were started on the Phase 2 over current relay; however, approximately 20-30 seconds after the paddle was inserted and prior to the apprentice electrician installing the glass cover, the 186-2 Lockout Relay tripped, causing isolation of bus 1A2. The electricians stopped work and contacted the Shift Manager.

In the Control Room, at 1125, annunciator 1C08 B (A-9) "Bus 1A2 Lockout Trip or Loss of Voltage" alarm was activated. Loss of 1A2 switchgear resulted in the loss of 'B' Reactor Feed Pump and 'B' Condensate Pump.

At 1125, a manual reactor scram was inserted due to Reactor Pressure Vessel level approaching 170 inches (automatic scram level). The scram went to completion. After the scram, vessel level rose to 211 inches, causing the 'A' Reactor Feed Pump to trip. Before the 'A' Reactor Feed Pump could be restarted and adequate feed flow established, another scram signal was received when reactor level dropped below 170 inches. Control rods were already inserted and isolations had been completed prior to this second scram signal being received. The Reactor Operator successfully restored level using the Feedwater Regulating Valve as directed by the Operations Shift Manager. The reactor was brought to MODE 3.

Notifications were made under 10 CFR 50.72(b)(3)(iv)(A) and 50.72(b)(2)(iv)(B) on April 2, 2007 and are listed as event number 43271.

II. Assessment of Safety Consequences:

This event was a lockout of the 1A2 bus and subsequent manual scram from 98% power (1880 MWth). The manual scram was inserted in anticipation of an automatic scram on low reactor water level. The low reactor water level was due to the trip of a feedwater pump and a condensate pump. The feedwater and condensate pump trips were due to the lockout of the 1A2 4160V AC electrical bus.

A trip of one feedwater pump is an anticipated operational transient described in Updated Final Safety Analysis Report (UFSAR) 15.1.7.4. As described in UFSAR 15.1.7.4, the expected plant response to a trip of one feedwater pump is a reactor scram and both HPCI and RCIC initiate on low-low reactor water level (119.5"). The actual plant response for this scram was much milder and clearly bounded by the trip of one feedwater pump event described in UFSAR 15.1.7.4.

Therefore, the plant shutdown did not result in any radiological or nuclear concern which would impact the health and safety of the public.

This event did not result in a Safety System Functional Failure.

III. Cause of Event:

An investigation into this event was completed under Root Cause Evaluation (RCE) 1065.

<u>Overall RCE Conclusions</u>

The human performance investigation conducted did not identify any discrepancies or inappropriate actions on the part of those involved in the field work or the control room crew on the day of the event.

Although not specifically targeted, equipment failures (relays, wiring, and panel configurations) were investigated. For the logic to be met (1A2 bus lockout) one of eight relays, or the lockout relay would have had to change state. Relays were sent off to an independent lab for testing. Detailed troubleshooting activities did not identify any source or cause that was consistent with or leading to the event. Given the fact that no equipment failures were identified, a human performance event (bumping or error) would be the likely cause of the actuation.

The RCE identified the main contributor of the event to be an organizational failure. Specifically, the organization failed to recognize the risk associated with performing the maintenance on-line without putting physical and process protections in place. Contributing to this cause was the successful performance of this maintenance on several previous occasions.

<u>Root Causes</u>

An organizational failure, which allowed work on a system with the risk potential for plant impact is considered to be the cause of this event. The failure was the lack of the organization to recognize the risk and place physical and process protection associated with performing the maintenance on-line. Contributing to this cause was the successful performance of this maintenance on several occasions previously. A human performance event (bumping or error) would be the likely cause of the actuation.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

IV. Corrective Actions:

Corrective Actions to Prevent Recurrence (CATPRs)

Change work orders / model tasks for bus relays associated with 1A1, 1A2, 1A3, and 1A4 to be done in a refueling outage mode. This corrective action should only be undone if a design / process change is made to facilitate work on-line without risk to tripping the lockout relays.

V. Additional Information:

Previous Similar Occurrences:

From LER review over the previous 5 years, the following four similar occurrences were identified in:

LER 2007-006 - Reactor Shutdown as a Result of a Chemistry Excursion
LER 2006-005 - Reactor Scram During Main Turbine Testing
LER 2003-005 - Unplanned Manual Reactor Scram due to High Reactor Coolant Conductivity
LER 2003-006 - Unplanned Manual Reactor Scram due to Degrading Condenser Vacuum

EIIS System and Component Codes:

EB - Medium-Voltage Power System-Class 1E

Reporting Requirements:

This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A).